FusionCanada

Bulletin of the National Fusion Program

Issue 27, March 1995

In this Issue

ISSN 0835-488X

ITER SITING

ITER Reactor Siting: Special Group Reviews Siting Criteria

At the request of the ITER Council, a multinational Special Review Group is now reviewing "the technical, social and safety and environmental requirements for siting ITER". The Group's first meeting was held February 14 - 16, 1995 in Naka, Japan. Dr. Robert Aymar, Director of the ITER Joint Central Team (JCT), and five senior JCT members, attended the meeting to present the Special Review Group (SRG) with current technical information on the ITER design, and on other technical and safety information related to siting requirements.

ITER is the International Thermonuclear Experimental Reactor being designed jointly by Europe, the USA, Japan and the Russian Federation (the 'four parties' to ITER). There is much interest among the ITER parties concerning the criteria to be used for selecting a construction site for ITER.

The SRG represents the four ITER parties; each party can nominate up to four Group members. Additional experts are also invited. The SRG was called into being at the 6th ITER Council meeting, held last year. It is chaired by Dr. Ken Tomabechi (JAERI, Japan), and will report its findings to the ITER Council later this year.

The SRG will consider technical information provided by the ITER JCT, including the ITER design as it currently stands, and other documentation including:

- General Design Requirements.
- General Safety and Environmental Design Criteria.
- Preliminary ITER Site Requirements and Site Design Assumptions.

Most discussions at Naka centred on the last of the three documents, particularly on its scope.

Contact persons for the SRG are:

European Union: J. Darvas (European Commission).

Japan: S. Matsuda (JAERI).

USA: W. Marton (Dept. of Energy).

Russian Federation: V. Korjavine (Minatom).

Tony Natalizio of the Canadian Fusion Fuels Technology Project, and Dr. T. Inabe of Japan, were invited as experts to the SRG meeting. Mr. Natalizio was invited as an expert on nuclear safety.

TdeV Tokamak
CCFM - Centre canadien de fusion magnétique

Major Upgrade for TdeV Tokamak

Work on a major upgrade of the TdeV tokamak will begin in mid-summer this year. The work should last about one year.

The TdeV upgrade will include installation of completely new divertors of advanced design and high power-handling capacity. After the upgrade TdeV will have new upper and lower divertors, and so will be capable of single null operation with either divertor, or double null operation using both.

The new divertors will be versatile, and capable of operating in different modes by changing their magnetic geometry (see diagram). Up to 8 independently controlled power supplies will shape the main plasma and magnetic geometry in the divertors, by controlling currents in the divertor coils and TdeV's poloidal field coils.

The different modes of divertor operation are basically these: 'slot' divertor and 'closed' divertor (each with either open throat or tight throat), and a divertor optimized for negative electrical plasma biasing. In essence, these geometries differ in very controlled alterations of the strike position and breadth of the plas-

continued inside

Major Upgrade for TdeV Tokamak

continued

ma impacting on the divertor plates. They also involve variable magnetic constriction of the divertor throat (inlet area).

The chief mission of the upgraded machine will be to explore divertor physics and test divertor plate materials at reactor-relevant power deposition levels. The new divertors will have entirely new divertor plates and supporting structures, and will be electrically insulated so that the main plasma can be biased via the divertor plates. These plates will have much enhanced heat absorption capacity, up to 12 MJ per tokamak shot, depending on operating scenarios.

Divertor power handling capacity will be reactor-relevant, so that experimental data can be applied to fusion reactor design. Power deposition rates (P/R) on the divertor plates may be up to 3 $MW.m^{-1}$.

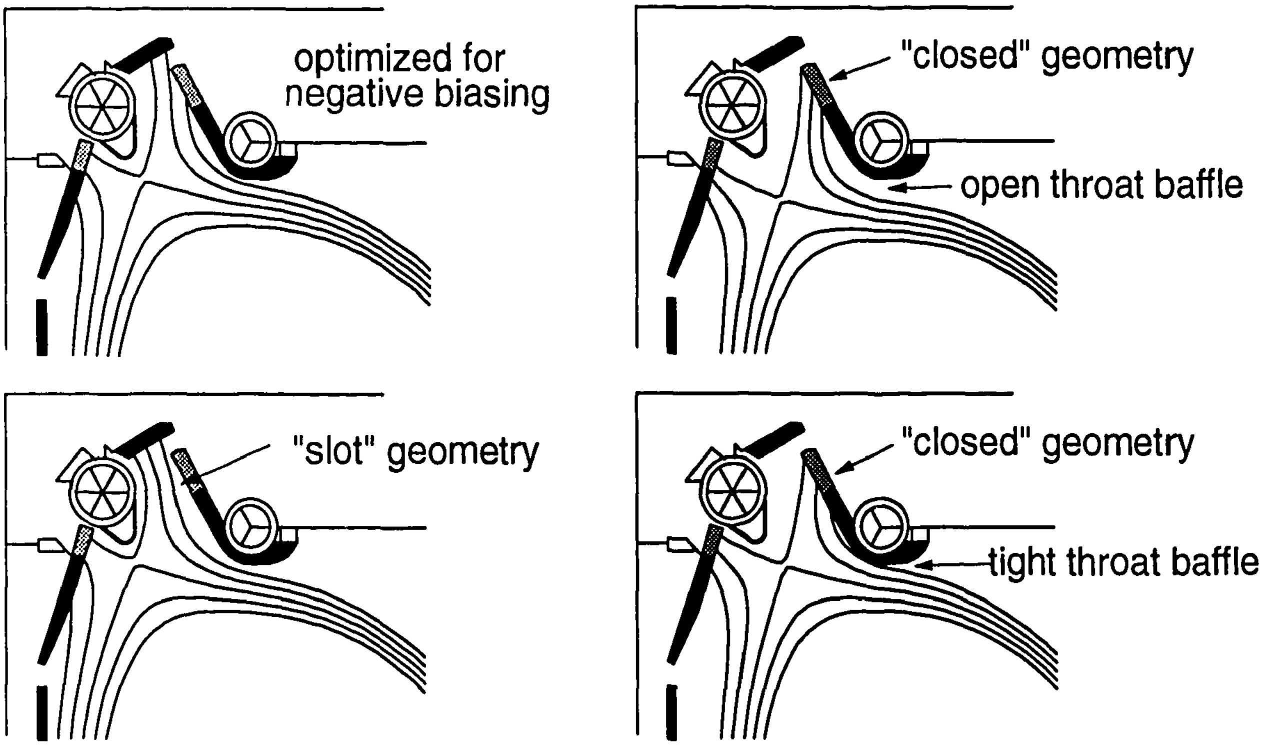
The main TdeV plasma (R ~ 0.83 m, r ~ 0.22 m) will be a little smaller than it is now, with a more triangular cross-section ($\delta < 0.5$). Toroidal field will be 1.95 tesla max., sustainable for 10 seconds.

Comparison of different divertor plate materials will be important. The new upper divertor will have carbon fibre composite divertor plates, to test low atomic number (low-Z) materials in high-power divertor service. The new lower divertor will be fitted with molybdenum plates, to test high-Z materials under the same conditions. At present, without substantial same-conditions data, debate about the benefits of low-Z plates versus high-Z plates remains unresolved. TdeV will be able to switch readily between upper and lower single null operation, to try tests of both.

The new power supplies, with a new plasma control system, will provide precise control of the position of the plasma strike point inside the divertors, and of the divertor throat baffling for particles.

More cryogenic vacuum pumps will be fitted, so that the lower divertor can be pumped, as is the upper divertor now.

TdeV Flexible Divertor Geometry



Typical modes of operation expected for the upgraded divertors to be installed on TdeV during the major upgrade that will start later this year. These illustrate the flexible throat baffling and strike point positioning in the outer divertor. The inner divertor will function as an open divertor.

TdeV Tokamak
CCFM - Centre canadien de fusion magnétique

Recent work on TdeV Tokamak

Plasma detachment in divertors

In March, the main research theme on the TdeV tokamak was the phenomenon of plasma detachment in TdeV's upper, outer divertor.

Since the re-start of TdeV, after a two-month winter maintenance shutdown, the CCFM team has chiefly explored divertor behaviour against variations in numerous parameters including main plasma conditions, electrical plasma biasing via divertor plates, and lower hybrid radiofrequency RF power injection (heating and current drive). Diagnostic instrumentation of the upper divertor was significantly improved during the winter shutdown, with the main aim of improving measurements of power balance within the divertors, and of power distribution between divertors and main plasma, under varying operating conditions.

Detachment of divertor plasmas on TdeV was first investigated quantitatively last year. It is being systematically characterized this year in a campaign scheduled to end in summer 1995. After that, TdeV will be shut down for a major upgrade (see separate article).

Density dependence

Like others, CCFM researchers have found a definite relationship between main plasma density and the degree of detachment of divertor plasmas from the divertor plates. Over a main plasma line average density (n_e) range of 3 - 8 x $10^{19}.m^{-3}$, detachment become increasingly apparent for n_e values over 5 x $10^{19}.m^{-3}$. This is not a threshold value, but more a point where plasma detachment noticeably increases versus n_e. It seems easier to detach divertor plasmas with plain ohmic discharges in TdeV, rather than when adding lower hybrid RF auxiliary heating. Indeed, detached plasmas can sometimes re-attach to the divertor

Plasma detachment in Divertors

Divertors are devices intended mainly for permanently removing impurities - helium, oxygen, metal ions and other species - from tokamak plasmas. Collectively, these particles - including a large flux of hydrogen fuel ions from the main plasma - carry a great deal of energy. Plasma-facing divertor components could thus receive great particle fluxes and heat loads that together would rapidly erode most materials, whether new or traditional.

Particles leaving the main tokamak plasma ('crossing the magnetic separatrix') are guided by shaped magnetic fields into a special volume near the main plasma (the divertor). Here, they become electrically neutralized by collision and charge exchange, with either divertor surfaces or neutral gas. Neutralized particles are then pumped out of the divertors - to permanently remove them - through vacuum pumps (TdeV uses cryogenic divertor pumping). Continuous vacuum pumping (exhaust) of the divertors should maintain main plasma purity, by counterbalancing the inflow (to the main plasma) of impurities from normal plasma-surface interactions inside the tokamak.

Because the particles swept into a divertor are already ionized, a hot plasma exists in the divertor. For a particle, neutralization is not granted automatically on entry. Plasma collected by the divertor is magnetically directed to strike a neutralizer plate, and so deposit considerable energy there. Without some form of intervention, enormous heat and particle loads - encouraging material erosion - can be placed on divertor components receiving the particle impacts. For a fusion reactor, projected surface heat loads are in the several MW/m^2 range, enough to rapidly erode the most resistant of structural materials.

By manipulating the tokamak's magnetic and plasma conditions, the bulk of the divertor plasma can be encouraged to move away - detach itself - from the surfaces of the divertor plates. This can reduce the power deposited on these plates, because incoming particles (from the main plasma) will tend to interact first with the detached plasma as they arrive in the divertor, instead of depositing most of their energy on the divertor structure. The detached plasma should then re-radiate and re-distribute its deposited particle energy, so that less heat impinges on the divertor plates. For example, some heat might be re-radiated back to the outer divertor components, where active cooling with water flow can remove it.

Detachment of divertor plasmas from the divertor neutralizer plates is possibly caused by buildup of a 'cushion' of neutral particles near the divertor surfaces.

BREEDER MATERIALS

3-D Tritium Mapping of Irradiated Lithium Zirconate Pebble Beds

Chalk River Technique

The Tritium Group at AECL's Chalk River Laboratories is using a technique that can produce a three-dimensional (3-D) profile of residual tritium levels in test beds of lithium ceramic breeder pebbles, after their irradiation in reactors. In-reactor tests of lithium ceramics are made to investigate matters including tritium release rates, ceramic material properties and tritium retention. The new 3-D tritium mapping technique is being used on a bulk cylindrical pebble bed of 1.2 mm diameter lithium zirconate (Li_2ZrO_3) pebbles made at Chalk River, and irradiated to high lithium burnup (5.2%) in the FFTF reactor at Hanford, Washington.

It is important for fusion breeder blanket designers to understand tritium retention in irradiated breeder ceramics; factors affecting tritium retention include lithium burnup, neutron spectrum, breeder temperature, and ceramic material structure and phase changes. In a fusion reactor blanket, all these factors will vary with time and with position in the blanket structure.

The FFTF Li_2ZrO_3 pebble test bed sample, provided by Chalk River, was a cylindrical capsule 13 mm dia. x 100 mm long, containing 30 grams of Li_2ZrO_3 pebbles, each 1.2 mm diameter.

The new tritium mapping technique involves injecting a liquid resin into the intact capsule, and after the resin solidifies, cutting away the capsule walls to leave a solid cylinder (or 'slug') containing the Li_2ZrO_3 pebbles fixed in a solid resin matrix. This composite cylinder preserves the spatial position of individual ceramic pebbles. Discs, or wafers, are then cut from the composite slug, at different points on the long axis of the cylinder. Each wafer is marked with a grid, for mapping, and then cut into small pieces. Tritium retained in the pebbles fixed in each small piece is measured by:

- first dissolving away the resin from the pebbles with a solvent,
- then dissolving the pebble fragments in strong acid.
- Then, the acid is distilled, and the tritium concentration in the acid distillate is measured by scintillation counting, to reveal the tritium content of pebbles in each piece of the wafer.

In this way, a 3-D map of tritium retained in the lithium ceramic pebble bed can be constructed.

Two wafers have been analyzed, so far, from the FFTF Li_2ZrO_3 pebble test bed. Measured tritium retention in the Li_2ZrO_3 varied from 0.02 ppm (by weight) to 0.36 ppm. More wafers are being analyzed.

Other groups at Chalk River will examine the fine structure of the irradiated Li_2ZrO_3 pebbles, with scanning electron microscopy, X ray-diffraction, nuclear magnetic resonance and other techniques.

The FFTF irradiation of the zirconate pebble bed was done under phase II of the BEATRIX-II breeder materials program, a joint Canada/USA/Japan breeder R&D program conducted under the auspices of the International Energy Agency. All irradiated zirconate analysis data will be shared with the USA and Japan as part of BEATRIX-II.

More information: Joan Miller, Tritium Group, AECL Chalk River (613) 584-3311, ext. 3277, Fax (613) 584-4445.

INTERNATIONAL

Joint CNS/FPA Fusion Symposium

Canadian Nuclear Society
Fusion Power Associates (USA)
Montréal, Canada, September 6-8, 1995

This event combines the annual technical meeting of the Canadian Nuclear Society's Fusion Committee, and the annual general meeting and technical symposium of the USA's Fusion Power Associates. The joint event will be held September 6-8 this year at the Radisson Hotel in downtown Montréal.

The theme of the meeting is: ***Status and Prospects for Fusion Power.***

Technical Sessions are on Thursday September 7 and Friday September 8. Delegates can visit the TdeV tokamak site, just outside Montréal, on the Friday afternoon. A reception and early registration will be held the evening of September 6.

Fusion Power Associates is an educational and research foundation established to foster the timely development and acceptance of fusion energy. The Canadian Nuclear Society is an association of professional engineers and scientists from universities, utilities and industry.

In Canada: Registration information from: Sylvie Caron, Canadian Nuclear Society. Phone: (416) 977-7620, Fax: (416) 979-8356. Program information: Guy LeClair, CCFM, (514) 652-8743, Fax (514) 652-8625.

In the USA, contact: Ruth Watkins at Fusion Power Associates. Phone: (301) 258-0545, Fax: (310) 975-9869, e-mail: 72570.707@Compuserv.com.

The 1.3 MW lower hybrid radiofrequency system will be retained, with a modified antenna. It is planned to supplement the lower hybrid system in 1988 with about 1 MW of electron cyclotron radiofrequency heating at 110 GHz.

Mission elements

In summary, the main elements of the mission for the upgraded TdeV are these:

- Study radiative divertor scenarios.
- Compare low-Z and high-Z divertor plates.
- Use variable divertor geometry (see diagram) and better heat handling capacity in high-power divertor studies.
- Study biasing for divertor heat flux and particle control.
- Improve magnetic geometry for H-mode and high ß operation.
- For single-null and double-null geometries, compare particle and heat exhaust via divertors.
- Study plasma current profile control and its effects on containment.

Upgrade planning is well advanced, and substantial amounts of equipment are being procured.

Réal Décoste is in charge of overall upgrade planning. Guenther Pacher of CCFM has special responsibility for the new divertor design.

More information: Réal Décoste (514) 652-8715, e-mail = decoste@toka.ireq-ccfm.hydro.qc.ca. Guenther Pacher (514) 652-8882, e-mail = pacherg@toka.ireq-ccfm.hydro.qc.ca. Fax (514) 652-8625.

Fusion Notes

CFFTP supplies tritium scavenger equipment for OMEGA Upgrade Laser at University of Rochester. The 60-beam OMEGA Upgrade laser at Rochester will irradiate hollow plastic microballoon targets containing tritium, during a program that will start this year. CFFTP is designing and supplying equipment to recover tritium released into the OMEGA Upgrade laser target chamber, when the laser implodes microballoon targets containing tritium. Under another CFFTP-Rochester agreement, Ontario Hydro Technologies is assessing the rate of diffusion of deuterium-tritium (DT) gas through the wall of aluminum-coated microballoon laser targets. Hollow laser targets are charged with DT fuel gas by diffusion. *More information: Ron Matsugu, CFFTP, (416) 855-4717, Fax (416) 823-8020.*

CFFTP and CCFM commit to more Compact Toroid Fueller work. The compact toroid fueller (CTF) now installed on the TdeV tokamak will be rebuilt to penetrate more intense magnetic fields on a tokamak. Results with the present CTF in 1994 were good - the CTF successfully injected compact toroid plasmas into TdeV, without main plasma disruptions, at toroidal fields up to 1.4 tesla. Over the next year the CTF and its power supplies will be upgraded to penetrate 2 tesla fields. The redesign and upgrade work is a joint project between CFFTP and CCFM. UC Davis and University of Saskatchewan are collaborators. *More information: Roger Raman, CCFM, (514) 652-8859, Fax (514 652-8625.*

STOR-M tokamak to starts up again. Variable-angle Compact Toroid Fueller fitted. The University of Saskatchewan's STOR-M tokamak will be restarted soon. It has been fitted with a new, flexibly-mounted compact toroid Fueller (CTF) for injecting small, dense hydrogen plasma toroids. The STOR-M toroidal magnetic field is also increased. The new CTF can swivel through 140 degrees in the STOR-M equatorial plane to inject CTs against the plasma current, or with the plasma current flow. Plasma current drive with CTs will be explored. Major changes to the STOR-M vacuum vessel were needed to fit the CTF, so STOR-M was shut down for some months. The tokamak now has an improved power supply, and a number of new microwave diagnostics in the 30 - 75 GHz frequency range. For the next year, the experimental program for STOR-M will concentrate on four main themes:

- Variable angle compact toroid injection
- AC tokamak operation
- Anomalous transport in tokamak plasmas
- Control of MHD activity

More information: Prof. Akira Hirose, Physics Dept., University of Saskatchewan (306) 966-6414, Fax (306) 966-6400, e-mail = hirose@sask.usask.ca.

DIVIMP Divertor code. This impurity transport code, from UTIAS, has been requested by staff at TdeV tokamak, and at the DIII-D tokamak in San Diego. (See FusionCanada No. 26, November 1994 for article on DIVIMP). Most of the world's divertor-equipped tokamaks are now using the code or have access. *More from Peter Stangeby, UTIAS. Phone/ fax (416) 667-7729. e-mail pcs@starfire.utias.utoronto.ca.*

plates when RF heating is switched on. With the present divertor design, plasma biasing via the divertor plates has no marked effect on detachment phenomena.

Predictability

Divertor behaviour remains difficult to predict with existing modelling codes, as observers everywhere agree. Density and temperature profiles in divertors do not agree well with code predictions, indicating that the underlying physics of divertor behaviour is yet to be fully learned. In TdeV experiments where detachment occurs, plasma density at the divertor plate surface decreases as detachment from the plates progresses. Existing codes would in general tend to predict otherwise, although the various codes do not always agree among themselves. Like many other sites, CCFM uses the Braams B2.5 code (CCFM lead investigator - Richard Marchand) and the EIRENE code (CCFM lead investigator - Magdi Shoucri) to model divertor behaviour.

The enhanced divertor diagnostics installed recently will provide more detailed profiles of plasma temperature and density, and of power deposition. Added diagnostics include more thermocouples, more flush mounted Langmuir probes, increased coverage with bolometry cameras, and reinstallment of video cameras and spectroscopy instruments for directly examining - in cross-section views - the divertor plasmas, at a tangent to the divertor plates. An infra-red camera observes temperature profiles on the plates.

Data from the present campaign are still being collected and digested. Barry Stansfield and Bernard Terreault coordinate divertor data acquisition and analysis.

Further information: Réal Décoste, Operations Director (514) 652-8715. Richard Marchand (514) 652-8866. Magdi Shoucri (514) 652-8723. Bernard Terreault (514) 652-8693. Barry Stansfield (514) 652-8735. FAX [all persons] (514) 652-8625.

National Fusion Program

Director, *Dr. David P. Jackson*

The National Fusion Program (NFP) co-ordinates and supports fusion development in Canada. NFP was established to develop Canadian fusion capability, in industry and in research and development centres NFP develops international collaboration agreements, and assists Canadian fusion centres to participate in foreign and international projects.

NFP is managed for Canada by Atomic Energy of Canada Limited. Federal funding is provided by Natural Resources Canada through the Panel on Energy Research and Development.

'FusionCanada' Bulletin

'FusionCanada' is available free to interested persons. It is published four times each year, in French and English editions. Write to NFP Office, 'Bulletin Subscriptions' (see Contact Data). Please specify French or English edition, (or both if desired), and number of copies if several are required.

Editors are invited to freely use or reprint items from 'FusionCanada'. Please credit 'FusionCanada' and National Fusion Program of Canada. Please also send to the NFP office a copy of the publication, or a proof or copy of the printed piece.

Contact Data

National Fusion Program

National Fusion Program
AECL
Chalk River Laboratories
Station E4A
Chalk River, Ontario
Canada K0J 1J0

Program Office: (613) 584-8036
Fax: (613) 584-4243

Dr. David Jackson
Director – National Fusion Program
(613) 584-8035

Dr. Charles Daughney
Manager – Magnetic Confinement
(613) 584-8037

Dr. Gilbert Phillips
Manager – International Program
(613) 584-8038

Dr. William Holtslander
Manager – Fusion Fuels
(613) 584-8039

CCFM
Centre canadien de fusion magnétique

CCFM
1804, montée Ste-Julie
Varennes, Québec
Canada J3X 1S1

Dr. Richard Bolton
CCFM Director-General
(514) 652-8701

Dr. Réal Décoste
CCFM Director-Operations
(514) 652-8715

Dr. Brian Gregory
CCFM Director-Research
(514) 652-8729

Secretariat: (514) 652-8702
Fax: (514) 652-8625

CFFTP
Canadian Fusion Fuels Technology Project

CFFTP
2700 Lakeshore Road West
Mississauga, Ontario
Canada L5J 1K3

CFFTP Program Manager
Dr. Donald Dautovich
(905) 855-4700

Enquiries: (905) 855-4701
Fax: (905) 823-8020

FusionCanada Office

Macphee Technical Corp.
80 Richmond Street West
Suite 1901
Toronto, Ontario
Canada M5H 2A4

Telephone: (416) 777-1869
Fax: (416) 777-9804

PERD

Ce Bulletin est aussi disponible en français

Printed in Canada on recycled paper